Third United Nations International Conference on the Peaceful Uses of Atomic Energy

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Abstract

DEVELOPMENT OF NUCLEAR FUEL INSPECTION TECHNIQUES IN JAPAN

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Inspection of nuclear fuels has a great significance from standpoints of securing the safety and economics of the fuels. Since the first reactor in Japan (JRR-1) came into operation in 1957, first inventory fuels for reactors have been imported, though fuels for several critical or sub-critical assemblies were produced in this country. It was JRR-3's second inventory fuels (natural uranium metal clad with aluminum) which were manufactured for the first time in domestic plants during 1963-64 and the Mitsubishi Group's research reactor fuels (enriched uranium oxide with aluminum sheath) followed.

In order to establish the inspection techniques for those home-made fuels, research and development works have been carried out among Japan Atomic Energy Research Institute (JAERI), Atomic Fuel Corporation (AFC) and private industries. Non-destructive testing techniques thus developed so far include: (1) application of ultrasonics for detecting internal defects in uranium billets, (2) radiographic measurement of aluminum cladding thickness, ultrasonic bonding test and determination of α contamination degree by ionization chamber device, which were developed especially for uranium metal fuels of JRR-3 and (3) eddy current test for sheath tubing of Mitsubishi Group's Reactor fuels. They are discussed in this report with emphasis upon the newly-developed or modified NDT techniques studied in Japan. Works have also been going on for other inspection techniques related to the other types of fuels that are considered to be used in this country, which will be briefly surveyed too.

There has been a committee in AFC comprised of specialists from fuel manufacturers, users and academic circles, which was established for the development of inspection techniques for home-made fuel elements as well as for imported fuels. Various other problems in the inspection of nuclear fuels have also been dealt with. The first effort has been made to obtain a criterion for the evaluation of welding practice for fuel rod end caps. Mechanical tests including tension and creep test, thermal cycle test as well as corrosion test were carried out on specimens of different designs which were so prepared as to have defects of known grade in the welded zone in order to determine the allowance limit of the types and grades of the defects there.

Standardization of fuel inspection techniques was the second problem: for example, in the measurement of density and grain size, method for thermal cycle test for uranium metal fuel core, measurement of particle size and O/U ratio of UO_2 powder, porosity and density of UO_2 pellets as well as in X-ray radiography of end cap welded zones.

Discussions were also made to find a suitable standard for the degree and range of application of various inspection methods for the respective research reactor fuels: specification of the fuel, operational conditions of the reactor and the experiences of manufacturing the similar type of fuels are among the important items considered. Eight research reactors already installed as well as other reactors yet on the planning stage were the subjects of consideration.

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IRRADIATION TESTS OF UO_2 FUEL: RADIOGRAPHY WITH BETATRON AND NEUTRON; RELOCATION OF FISSION PRODUCTS AND PLUTONIUM

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Natural uranium UO_2 pellets fabricated by the "roll method" were irradiated in GETR at high temperature (specimens S-1, S-2, temperature ~ 2000 °C at the surface and ~ 3000 °C at the center) and at lower temperature (specimens S-3, S-4, temperature ~ 900 °C at the surface and ~ 1500 °C at the center). Each of four sets with six pellets, 12 mm in diameter and 12 mm in length was put in a dual stainless steel capsule and were irradiated for one and two months, at respective temperature. Out of many findings two remarkable results are reported here.

RADIOGRAPHY WITH BETATRON AND NEUTRON

By taking advantage of the penetrating power, low scatter, and extremely fine focus of betatron X-rays, a technique taking clear and high resolving radiographs of a fuel has been developed.

Distance between the film and the specimen was 50 - 100 cm and a steel plate interposed between them. The distance between the film and the betatron focus was 100 - 200 cm. The intensity ratio of betatron X-rays to specimen radiation which determines clearness of radiograph became very

large (e. g. > 1000) as compared with a conventional X-ray radiograph in which film specimen distance was very small and because penetration of 15 MeV X-rays is larger than fuel gamma ray by a factor of 10 with a 7 cm iron plate.

Betatron radiographs of S-1, and S-2 show that the inner capsules were molten down except at both ends, and pellets themselves were united completely into a single rod with a big void along the central axis of the rod. Those of S-3, and S-4 shown that inner capsules were sound and each pellet cracks, especially outer half of the top pellet was severely deteriorated. A neutron radiograph of the specimen by our research reactor TTR - 1 neutron flux also gives a clear image especially of the void. Results of pinhole camera autoradiography of the specimens are far inferior to the above.

It is concluded that betatron radiography is especially effective for non-destructive testing for fuel when it has such high activity as more than several thousand MWD per ton.

RELOCATION OF FISSION PRODUCTS AND PLUTONIUM

The specimens mentioned above, of about 0. 05 % (S-1) and 0. 1 % (S-2) burnups were examined about distribution of fission products (Ce, Ru, Zr-Nb and Kr) and plutonium along the length and also at cross sections of the rods. Distribution of Ce, Pu and Ru were found fairly normal while the concentration of Cs was quite high at both ends of the rod where the temperature was lower than the central part. Zr-Nb was found more concentrated in the inner part. About two percent of Kr-85 was found to be remained in S-1. Distribution of specific activity throughout the rods was quite irregular having wide depleted sections.

On the other hand, the distribution of fission products and Pu in specimens which were irradiated under lower temperature were quite uniform, which suggest that migration of fission products and plutonium should be considered in UO_2 fuel which are irradiated near the melting point.

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On the Thermal Fatigue Behavior of Fuel Sheath Material subject to Constrained Deformation caused by Core Material.

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Sheath and core materials bonded together in a fuel element into a single body are nevertheless subjected to different conditions of temperature and atmosphere. When the rigidity of the sheath material is lower than that of the core material, the sheath would yield to dimensional changes of the core during use in reactor and undergo constrained deformation. Hence, for fuel element design, safety analysis and use, it should be very important to study this constrained deformation from the standpoint of thermal fatigue behavior.

For the present study a universal thermal fatigue testing machine of the authors' original design was used, this machine can be utilized not only for investigating such constrained deformation but also other thermal fatigue phenomena in general.

The experimental method is to fix the core and sheath separately, and have the thermal strain induced by the difference in thermal cycling temperature range accurately transmitted to the sheath, thus permitting the induced stress behavior, microstructure and failure to be observed.

1. Experiments on 2S Aluminum.

2S Aluminum kept at 75°C was subjected to constrained deformation caused by β-quenched uranium undergoing thermal cycling in the range 110°C- 550°C. It was found that the maximum value of the stress induced in the aluminum decreases gradually up to 40 cycles and then becomes constant. Furthermore the yielding stress of the aluminum decreases with increasing number of cycles with corresponding lower temperature of the uranium.

Constant strain cycling tests were also carried out at several rates

of increasing strain provoked by thermal growth of uranium. The results were as follows.

a. Increasing strain of any rate augments local deformation in aluminum.

b. The stress change in aluminum depends upon the rate of increasing strain.

c. Between number of cycles to fatigue failure, N, and rate of increasing strain, $\Delta\varepsilon$, there exists a relationship, $\Delta\varepsilon \cdot N^n = C$, where C and n are constant. These results are discussed from considerations of structural change in the aluminum.

2. Experiments on Magnox A-12 alloys.

Magnox A-12 alloy cycled between 150°C and 450°C was subjected to constrained deformation caused by uranium undergoing cycling between 190°C and 575°C, with the same cycling period for both materials. It was found that compressive stresses occurred in the Magnox at higher regions of the temperature range, and that lengthening the cycling period from 0.5 to 4 hr. per cycle promoted tensile creep and stress relaxation in the lower temperatures. Finally some ordinary thermal fatigue tests for Magnox were carried out in five steps of lower temperature limit ranging from 125°C to 325°C and with common temperature interval of 200°C. With higher temperature limits, for instance 525°C, the decrease of fatigue life was quite marked. From this the authors have concluded that in such cases the well-known Coffin's low was no longer satisfied, and that to explain the effect a concept involving thermal activation might be introduced for higher temperature ranges.

Appendix.

The authors have investigated deformation and changes in dimension, strength and microstructure during simple thermal cycling of various metals and alloys for reactor use, and many interesting characteristics have been newly found, some of which proved to be useful keys considering the basic concept of thermal fatigue behavior of the sheath materials named above.

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Abstract

DEVELOPMENT OF BONDING METHODS FOR GRAPHITE MATERIALS

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Graphite possesses good nuclear properties, and is resistant to heat and corrosion, which properties have caused this material of recent to be rapidly developed for nuclear uses. Graphite, however, is very difficult to fabricate and, particularly, bonding without affecting the above properties has presented many problems.

Since 1959 Tokyo University, Japan Atomic Energy Research Institute, Tokai Electrode Mfg. Co. and Nippon Carbon Co. have co-operated in a project for developing brazing materials and methods, and a number of notable results have been obtained.

1. Brazing materials

The authors have found a new brazing material of Ni-Fe-Ti alloy. The brazing temperatures possible with this material are 1300-1600°C, so that the brazed graphite joints are serviceable at temperatures of 700-1000°C. Tested for long periods at elevated temperature, the bond showed no indication of carbide formation to embrittle the bond.

Another form of the same brazing material developed by the authors is a paste obtained by mixing the metals in powdered state, using an organic compound such as styrene for binder. The advantage of this paste is the possibility of obtaining different kinds of composition to suit particular purposes, thus providing a very versatile brazing material. The binder decomposes by heat during brazing, but leaves little residual carbon, and being free of oxygen in its molecular structure it does not affect the bond.

2. Brazing methods

The authors have further contributed to the development of vacuum brazing, inert-gas-shielded arc brazing (Tig brazing), and resistance brazing. Other new techniques developed include reinforcement with metallic deposit beads and double-pass Tig brazing.

3. Bonding of metallic terminals on SiC electric heaters

SiC is a good heater material for in-pile experiments since it can provide high temperatures and high power density, at the same time permitting the use of a high voltage and low current system.

SiC heater terminals have in the past been fixed mechanically, which limited the temperature at terminal to 200°C.

The authors have found a new process by which titanium terminals can be bonded to SiC heaters with graphite as interspacer.

The terminals fixed by this process are capable of withstanding 1000°C.

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Abstract

COMMERCIAL TECHNICAL METHODS FOR ENRICHMENT AND PURIFICATION OF DEGRADED AND CONTAMINATED HEAVY WATER FOR NUCLEAR REACTORS

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Development research on heavy water reactors has been conducted in Canada, France, the United States, the United Kingdom, the Soviet Union, Sweden, etc., which has induced the economy of the heavy water reactor to be reviewed with renewed interest in recent years. Heavy water in a reactor is reduced in concentration or subject to contamination depending on the handling conditions. Economical enrichment and refinement of heavy water is a vital question in connection with the operation of heavy water reactors.

We made pilot plant tests on the non-diaphragm electrolysis process with special recovery apparatus, and the rectification process, and established new, workable techniques in both processes. We have completed new technology including necessary conditions to scale up apparatus (Japanese Patent No. 256416, Japanese Utility Models No. 714723 and No. 714724), the optimum cascade theory and economic evaluation. These processes are indispensable to improve the economy of the heavy water reactor. To prevent contamination and degradation of heavy water in treatment, the gas circulation and recovery system is of special design. The operating condition is very safe and the yield is about 100%. The distillation process consists of a sieve plate tower. The design is based on the ideal cascade theory, and the operation cost is very low. Domestically, we have rendered appreciable service in the operation of JRR-2 and JRR-3 of the Japan Atomic Energy Research Institute. New economic reconcentration and regeneration of heavy water is being carried out with a 5 T/Y plant.

The present report contains results of such experiment and design calculation as to a 30 T/Y plant capable of treating larger quantities of

heavy water. It also acknowledges that the aforesaid electrolysis process is economically advantageous for a capacity of less than 5 T/Y and that the rectification process for a higher capacity.

As to removal of impurities in heavy water, including oil and soluble ion, we have gained experience in highly efficient rectification by operating a pilot plant. We feel that we have established a low cost process for regenerating heavy water which has been used in a reactor.